NRC STAFF RESPONSE TO PUBLIC COMMENTS ON DRAFT REGULATORY GUIDE DG-1356: GUIDANCE FOR IMPLEMENTATION OF 10 CFR 50.59, "CHANGES, TESTS, AND EXPERIMENTS"

Federal Register 84 FR 25077 (May 30,2019)

On May 30, 2019, the NRC issued for public comment in the *Federal Register* (84 FR 25077) draft regulatory guide (DG), DG-1356, "Guidance for Implementation of 10 CFR 50.59, 'Changes, Tests, and Experiments'" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19150A630). Comments on the subject DG are available electronically at the U.S. Nuclear Regulatory Commission's (NRC's) electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can enter ADAMS, which provides text and image files of NRC's public documents.

NRC staff received 3 comment submissions on DG-1356, as listed below, from the following individuals or organizations:

Letter No.	ADAMS Accession No.	Commenter Affiliation	Commenter Name
1	ML19198A321	No Known Affiliation	Anonymous
2	ML19198A322	Entergy Nuclear Operations, Inc	Philip Couture
3	ML19198A323	Nuclear Energy Institute (NEI)	Stephen E. Geier

This document lists each public comment by letter and comment number. For example, Comment 3-2 would be the second comment provided in Letter No. 3 listed in the table above. For each comment, the NRC has repeated the comment as written by the commenter and then provided the NRC's response. In some instances, the comment was broken down into segments for clarity.

Comment No. 1-1

You should not issue your Appendix D until the investigation of the Boeing 737 crash is finished. Is the RG 1.187 for digital the same as the FAA digital guidance for airplanes? The NY Times article on June 1, 2019 states that regulators were left in the dark about a fundamental overhaul to an automated system that would ultimately play a role in two crashes. The regulators did not perform a safety assessment of a newer update of the computer controls in the planes. The RG 1.187 allows computer controls of old nuclear reactors without a regulator review at all. It would even allow failure of automated systems, and trust the power plant to calculate that accidents will not have bad consequences. The NRC should review and include the findings of the Government investigation of the FAA in RG 1.187. If nuclear automation could fail and cause accidents, it must be given to NRC for approval. Do not repeat what FAA failed to do.

The RG 1.187 needs to have plans for inspecting nuclear digital automation with RG 1.187. Please explain.

NRC Response

The NRC staff considers this comment to be addressing the underlying regulatory process established in 10 CFR 50.59, which allows for a licensee to make changes to its facility as described in the Final Safety Analysis Report (FSAR), as updated, without seeking an amendment from the NRC. Through 10 CFR 50.59, the NRC ensures that every nuclear power reactor licensee seeks a license amendment to request the NRC to authorize any change to its facility that is not encompassed within the existing NRC-approved basis for licensing the facility. The NRC conducts periodic inspections to monitor the effectiveness of licensee implementation of the 10 CFR 50.59 process and permanent plant modifications, which provides assurance the licensee has obtained the required license amendments. If the licensee has not sought an amendment, 10 CFR 50.59 requires the licensee to document its reasons for not doing so, i.e., why the change to the facility is already encompassed within the licensing basis stated in the FSAR. Further, in contrast to the regulatory process in 10 CFR 50.59, which addresses changes to existing licensed facilities, the FAA process at issue in the comment has to do with new approvals and thus is not a comparable regulatory process. Finally, NEI 96-07, Appendix D and the companion guidance in RIS 2002-22 Supplement 1, "Clarification on Endorsement of NEI Guidance in Designing Digital Upgrades in Instrumentation and Control Systems," "…is not directed toward digital I&C replacements of the reactor protection system, the engineered safety features actuation system, or modification/replacement of the internal logic portions of these systems (e.g. voting logic, bistable inputs, and signal conditioning/processing) because application of the guidance in this RIS supplement to such changes would likely involve additional considerations."

No change was made to the regulatory guide in response to the comment.

Comment No. 1-2

You need to do an environment impact assessment of digital changes that can allow errors in high radiation detectors needed for a LOCA. Your Appendix D says that radiation monitors can be installed without a low chance of failure. A different result does not happen because there is nothing in the safety analysis. Why would NRC allow struggling nuclear power plants to install this type of equipment and not even want to review it? Please explain. Radiation monitors are used by the State of Illinois to decide evacuation plans during a meltdown accident. The State should be notified of these changes of radiation monitors that are considered and be able to provide comments on environmental assessments and evacuation.

The NRC needs to notify the State that the nuclear power plant can calculate different results without NRC review because of the safety analysis. Did the NRC agree to the wording of the original safety analysis knowing computers would later be used? Did the State know this?

Please explain if automation with common failures be allowed if the toxic fuel can get closer to melting without NRC? Is there hearing rights if there is a chance of a new common failure with worst conditions inside the reactor than agreed to before with original equipment?

NRC Response

The NRC staff agrees with the comment that Appendix D, Revision 0, is incorrect to the extent that it asserts that malfunction results can only be different if there is an evaluation in a safety analysis. The NRC included an exception to address that issue in Section C.2.e of the draft regulatory guide: "Stating that there cannot be a different result when comparing to a preexisting safety analysis because none exists is not adequate to meet 10 CFR 50.59."

By letter dated, May 13, 2020 (ADAMS Accession No. ML20129J857), NEI submitted a revised Appendix D to address the issues described in Section C.2.e of the draft regulatory guide. The revised Appendix D corrected the issue identified by the comment. The NRC review of NEI's May 13 revised Appendix D, determined that Appendix D, Section 4.3.6, as revised, is acceptable and addresses the key point of comment No. 1-2. As a result, a change was made to delete the exception in Section C.2.e of the draft regulatory guide

The comment also appears to address radiation monitors used to measure sources of radioactivity release offsite during an accident, but these types of radiation monitors are not addressed in Appendix D. Rather, Example 4-19, which contains the only discussion of radiation monitors in Appendix D, discusses area radiation monitors that monitor areas within the nuclear power plant (e.g., rooms, hallways, etc.).

The comment also questions whether environmental review will be conducted and whether there are hearing rights. If 10 CFR 50.59 requires a licensee to seek a license amendment to authorize a particular change and the licensee submits a license amendment request, the NRC will perform an environmental review and offer an opportunity for a hearing. If 10 CFR 50.59 allows a licensee to make a change without seeking an amendment, there is no NRC action, and no environmental review is required, nor is there an opportunity for a hearing. As the NRC staff understands the example posited in the comment regarding "[worse] conditions inside the reactor" compared to the licensing basis existing before the change, a license amendment would be required (i.e., the existing safety analysis is not bounding for the proposed change); however, a reduction in margin, e.g., margin to fuel damage, is allowed via 10 CFR 50.59 if the licensing basis existing before the change describes conditions "worse" than those which could result from the change (i.e., the existing safety analysis remains bounding for the proposed change).

Comment No. 1-3

The NRC says that a common failure can be low enough with a quality assessment. Please explain the evidence and data you used for knowing when the chance of common failure is low enough.

The RG 1.187 says you reviewed international guidance for harmonization and found nothing useful. Please list all the guidance that was reviewed. We did a google search and found IAEA guidance. Does IAEA and other countries agree that common failure can be shown to be low enough with a quality assessment? Please explain.

NRC Response

From the context in this comment, the NRC staff understands that the commenter was referring to a "qualitative assessment," which is used to demonstrate that the likelihood of software CCF is sufficiently low. Whether a software CCF is sufficiently low is determined on a case-by-case basis using the guidance in NRC Regulatory Issue Summary (RIS) 2002-22, Supplement 1, "Clarification on Endorsement of Nuclear Energy Institute Guidance in Designing Digital Upgrades in Instrumentation and Control Systems," dated May 31, 2018 (ADAMS Accession No. ML18143B633). Section C.2.b, Software Failure, of RG 1.187 states, "RIS 2002-22 Supplement 1, should be used in conjunction with NEI 96-07, Appendix D to provide an acceptable technical basis to determine that the likelihood of software CCF is sufficiently low for the purpose of 10 CFR 50.59 evaluations."

The NRC has a goal of harmonizing its regulatory guidance with documents issued by the International Atomic Energy Agency (IAEA) to the extent practical. The NRC staff has reviewed the IAEA standards and guides and did not identify any documents with useful relevant information related to the topics in this RG.

No change was made to the regulatory guide in response to the comment.

Comment No. 2-1

Entergy Nuclear Operations, Inc. and Entergy Operations, Inc. (collectively referred to as “Entergy”) are providing this letter in response to the NRC request for comment on Draft Regulatory Guide, DG-1356. Entergy has been an active participant in the NRC and industry meetings regarding this topic and endorses the comments provided by the *Nuclear Energy Institute (NEI).*

NRC Response

The commenter provided no specific comment other than stating that it endorsed the comments on DG-1356 provided by NEI.

See NRC Response to Comments 3-1 through 3-8 below that address NEI comments on DG-1356.

Comment No. 3-1

B. Discussion, Background, Page 5, Paragraph 5

The draft guidance states, “NEI 96-07, Appendix D, does not replace or supersede NEI 01-01 either in whole or in part. Licensees have the option to use the 10 CFR 50.59 guidance in total in either NEI 01-01 or in NEI 96-07, Appendix D.” This is confusing because NEI stated its intent that, “The guidance in this appendix supersedes the 10 CFR 50.59-related guidance contained in NEI 01-01/ EPRI TR-102348, Guideline on Licensing of Digital Upgrades, and incorporates the 10 CFR 50.59-related guidance contained in Regulatory Issue Summary (RIS) 2002-22, Supplement 1, Clarification on Endorsement of Nuclear Energy Institute Guidance in Designing Digital Upgrades in Instrumentation and Control Systems.”

Recommendation:

Clarify that NEI 96-07, Appendix D supersedes the 10 CFR 50.59-related guidance contained in NEI 01-01/ EPRI TR-102348, Guideline on Licensing of Digital Upgrades. NEI will not be making further changes to update or maintain NEI 01-01. If NRC wishes to retain for licensees the option to use NEI 01-01, that can still be specified.

NRC Response

The NRC staff understands that NEI does not intend to make any further changes to update or maintain NEI 01-01. The NRC staff revised the wording in the regulatory guide to include the quote from Appendix D, stating: “The guidance in this appendix [NEI 96-07, Appendix D] supersedes the 10 CFR 50.59-related guidance contained in NEI 01-01/ EPRI TR-102348, Guideline on Licensing of Digital Upgrades, and incorporates the 10 CFR 50.59-related guidance contained in Regulatory Issue Summary (RIS) 2002-22, Supplement 1, Clarification on Endorsement of Nuclear Energy Institute Guidance in Designing Digital Upgrades in

Instrumentation and Control Systems." The NRC continues to endorse NEI 01-01; licensees continue to have the option to use NEI 01-01, as described in RG 1.187, Section C.2.a.

Comment No. 3-2

Section C.2.a, NEI 96-07, Appendix D Use

The draft guidance in C.2.a. is confusing and unnecessary.

Recommendation:

Section C.2.a. could be eliminated by revising the Section 2 introductory statement to something along the lines of: "The NRC staff evaluated NEI 96-07, Appendix D, as applied to digital modifications only. The NRC staff concludes that Appendix D provides an acceptable approach for the application of 10 CFR 50.59 guidance when conducting digital instrumentation and control modifications, subject to the following exceptions and additions:"

NRC Response

The NRC staff agrees with the comment. The wording in section C.2.a ("NEI 96-07, Appendix D Use") of DG-1356 was eliminated and a statement consistent with the provided recommended text was incorporated into the Section C.2 introductory statement.

Comment No. 3-3

Section C.2.b, Human-System Interface

The draft guidance states (in part), "However, including Human-System Interface (HSI) changes in the screening process is a change from the guidance contained in NEI 96-07, Revision 1, Section 4.2.1.2."

This statement is incorrect.

NEI 96-07, Rev. 1, Section 4.2.1.2 contains the following guidance:

"For purposes of 10 CFR 50.59 screening, changes that fundamentally alter (replace) the existing means of performing or controlling design functions should be conservatively treated as adverse and screened in. Such changes include replacement of automatic action by manual action (or vice versa), changes to the ***man-machine interface****, changing a valve from "locked closed" to "administratively closed" and similar changes."* ***[emphasis added]***

The concept of man-machine interface, now called human-system interface, was previously considered in NEI 96-07, Rev. 1, Section 4.2.1.2.

NEI 01-01, Section 4.3.4 also currently considers the human-system interface.

Recommendation:

Delete the subject sentence.

<u>NRC Response</u>

The NRC staff agrees with the comment. NEI 96-07, Revision 1, included the guidance that all HSI changes should automatically screen in. The subject sentence in DG-1356 was intended to communicate that the screening process in NEI 96-07, Revision 1, automatically screened in changes to HSI as adverse, but Appendix D does not automatically screen in changes to HSI, instead providing screening guidance specific to HSI. The sentence in Section C.2.b.i (previously Section C.2.b in DG-1356), "Human-System Interface," of the regulatory guide was changed in response to the comment. See NRC response to Comment 3-4.

<u>*Comment No. 3-4*</u>

Section C.2.b, Human-System Interface

The draft guidance states (in part), "Digital interfaces are fundamentally different from analog interfaces." This statement is contradictory to the 10 CFR 50.59 guidance currently endorsed by the NRC in NEI 01-01.

Originally (i.e., before NEI 01-01), NEI 96-07, Rev. 1, Section 4.2.1.2 contained the following guidance:

"For purposes of 10 CFR 50.59 screening, changes that fundamentally alter (replace) the existing means of performing or controlling design functions should be conservatively treated as adverse and screened in. Such changes include replacement of automatic action by manual action (or vice versa), changes to the ***<u>man-machine interface</u>****, changing a valve from "locked closed" to "administratively closed" and similar changes." [****<u>emphasis added</u>****]*

This guidance meant that ALL man-machine interfaces (now called human-system interfaces) MUST be considered ADVERSE (i.e., "screen in").

Then, NEI 01-01 was endorsed by the NRC and Section 4.3.4 contained the following guidance:

"It is important to note that not all changes to the human-system interface fundamentally alter the means of performing or controlling design functions. Some HSI changes that accompany digital upgrades leave the method of performing functions essentially unchanged. Technical evaluations should determine whether changes to the HSI create adverse effects on design functions (including adverse effects on the licensing basis and safety analyses)."

This guidance, which is currently endorsed, clearly states that the impact of a change to an HSI (i.e., a Human-System Interface) on a UFSAR-described design function <u>needs to be determined</u>. In other words, an HSI change no longer <u>automatically</u> becomes ADVERSE, or defaults to being ADVERSE.

The sentence proposed by the NRC (identified in the first paragraph above) overturns the guidance in NEI 01-01 and returns the guidance to that given in NEI 96-07. Furthermore, the intent of the guidance in Appendix D is to provide one type of technical evaluation that the 50.59 practitioner may use to determine the impact of an HSI change on a UFSAR-described design function.

If the proposed sentence is maintained as written, then there is no need for the guidance contained in Appendix D, Section 4.2.1.2 since ALL changes involving an HSI would need to be *considered ADVERSE.*

Recommendation:

Consider deleting the sentence entirely or at a minimum modifying the sentence to read: "Digital interfaces are <u>not necessarily</u> fundamentally different from analog interfaces." [<u>emphasis added</u> to highlight the suggested modification]

<u>NRC Response</u>

The NRC staff agrees with the comment that draft guide statement, "Digital interfaces are fundamentally different from analog interfaces," is unclear. Section C.2.b.i (previously Section C.2.b in DG-1356) was updated to explain some of the background behind the differences between NEI 96-07, Revision 1, and Appendix D on Human-System Interface (HSI) guidance for the screening process and to acknowledge that Appendix D includes guidance for screening changes to HSI.

<u>Comment No. 3-5</u>

Section C.2.c, Examples Illustrate Guidance

The draft guidance states (in part), "For example, the "Note" in example 4-19 of NEI 96-07, Appendix D states, "The acceptability of these new area radiation monitors will be dictated by their reliability, which is assessed as part of Criterion (ii), not Criterion (vi)." The NRC staff's position is that this note is potentially misleading as it could be read to mean that CCF of a proposed digital I&C modification is solely a reliability issue, applicable to Criterion (ii) and not Criterion (vi), when read within the context of the entirety of example 4-19."

This statement is a "comment," NOT an "exception."

Recommendation:

Delete the identified text since it is not an exception in the form of a "limit/restriction" on the use of the examples (as is done in the first two sentences).

<u>NRC Response</u>

The NRC staff agrees with the comment. The commenter's recommendation to delete the identified text from Section C.2.c of the draft regulatory guide is appropriate based on NEI's revised Appendix D submitted to the NRC on May 13, 2020, which revised the text of example 4-19.

A change to regulatory guide was made to delete the excerpt from Section C.2.c stated in the comment.

<u>Comment No. 3-6</u>

Section C.2.d, Software Common Cause Failures

This section is confusing. It could be misinterpreted to imply that NRC staff takes exception to all Appendix D language discussing software CCF except for language quoted directly from RIS

2002-02 Supplement 1.

Recommendation:

Section C.2.d. should either be deleted -or- it should be revised to contain the specific software CCF related text in Appendix D to which the staff take exception.

NRC Response

The NRC staff agrees with the comment, but not the commenter's recommendation. Section C.2.c (previously Section C.2.d in DG-1356) was revised to state that RIS 2002-22 Supplement 1, should be used in conjunction with Appendix D to provide an acceptable technical basis to determine that the likelihood of software CCF is sufficiently low for the purpose of 10 CFR 50.59 evaluations.

Comment No. 3-7

Section 4 of RG 1.187, Revision 1

Section 4 of RG 1.187 Rev 1 titled "Applicability to 10 CFR Part 50 Licensees other than Power Reactors" has been deleted from draft of Rev 2 with no apparent explanation. This is confusing and likely to be interpreted as effectively eliminating 10 CFR Part 50 Licensees other than Power Reactors from the scope of this RG. Some Part 50 Licensees other than Power Reactors need the guidance contained in Appendix D and this RG to fulfill their missions.

Recommendation:

The language of Section 4 of RG 1.187 Rev 1 should be re-included in Rev 2. Alternatively, the staff should state in the revised RG why it was removed and provide an analysis of the impact the change in regulatory guidance would have on affected Part 50 Licensees.

NRC Response

The NRC staff agrees with the comment and the recommendation to include the subject language. The subject language in C. Staff Regulatory Guidance, Section 4, of RG 1.187, Rev. 1, was added as C. Staff Regulatory Guidance, Section 5 of the regulatory guide as follows:

> 5. Applicability to 10 CFR Part 50 Licensees other than Power Reactors
>
> While most of the examples and specific discussion focuses on power reactors, 10 CFR Part 50 licensees other than power reactors may use the guidance contained in Revision 1 of NEI 96-07. However, certain aspects of the guidance discuss regulatory requirements that may not fully apply to these licensees (e.g., Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants").

Comment No. 3-8

Section C.2.e, Section 4.3.6 of NEI 96-07, Appendix D

In DG-1356, Section C.2.e, "The NRC staff takes exception to the application of the term 'safety analysis' to the criterion in section 10 CFR 50.59(c)(2)(vi) in lieu of the term 'FSAR (as updated)' *throughout NEI 96-07, Appendix D, Section 4.3.6." Section C.2.e further states, "The NRC staff's position is that where the criteria in 10 CFR 50.59 uses the term 'previously evaluated in the final safety analysis report,' it means the whole FSAR (as updated). Therefore, when applying the guidance in Appendix D, licenses should not limit their examination of the FSAR (as updated) to particular sections."*

The guidance proposed in NEI 96-07, Appendix D, Section 4.3.6, specifically the six step process for cases in which the qualitative assessment outcome is a failure likelihood of not sufficiently low, begins with identification of all functions that are directly or indirectly related to the proposed activity. Further, the guidance reiterates the expectation from NEI 96-07, Rev. 1 that all functions involved with the proposed activity are initially considered in the scope of review regardless of the level of direct description in the FSAR (as updated) or UFSAR. This is consistent with the NRC staff position that one must examine the whole FSAR (as updated).

However, because 10 CFR 50.59(c)(2)(vi) states, "Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report (as updated)," each of the involved functions must then be examined to determine which are design functions. That is, "malfunction of an SSC important to safety" has been defined in Definition 3.9 of NEI 96-07, Rev.1 as "the failure of SSCs to perform their intended design functions described in the UFSAR (whether or not classified as safety-related in accordance with 10 CFR 50, Appendix B)." From the discussion in NEI 96-07, Rev. 1, Definition 3.3, "Design functions are UFSAR described design bases functions and other SSC functions described in the UFSAR that support or impact design bases functions." This discussion continues, providing the definition of design bases function from Appendix B to NEI 97-04 as endorsed by Regulatory Guide 1.186. The NRC has previously endorsed all these definitions and related discussions of "design functions" and "design basis functions" in NEI 96-07, Rev. 1 and NEI 97-04, Appendix B.

The definition of "malfunction of an SSC important to safety" and the focus on design functions are a direct reflection of the 1999 rulemaking on 10 CFR 50.59, which was promulgated to address the uneven application of the rule to licensees with UFSARs of varying level of detail. The associated design functions are described in licensees' UFSARs, and both NEI 96-07, Rev. 1 and Appendix D provide guidance to ensure that these design functions are properly treated. DG-1356 is silent on the regulatory foundation for "malfunction of an SSC important to safety" as there is no mention of NEI 96-07, Rev. 1 Sections 3.9 and 3.3, or RG 1.186.

With a "malfunction of an SSC important to safety" being "the failure of SSCs to perform their intended design functions described in the UFSAR," it is clear that the result of the failure to perform a design function is the focus. Returning to the discussion in NEI 96-07, Rev. 1, Definition 3.3, the connection between design functions and design bases functions is described. NEI 96-07, Appendix D, Section 4.3.6, provides guidance on taking each design function through a process to determine the result of a failure to perform that design function.

NEI 96-07, Appendix D, Section 4.3.6 reasonably interprets the term "different result" in criterion 6 to mean "different safety analysis result." While DG-1356 takes exception to this position, it points to no agency guidance offering a contrary interpretation, nor does it demonstrate that NEI's position is unreasonable or would result in any safety issues. On the other hand, NEI's proposal has the advantage of allowing licensees to use the endorsed definition in NEI 96-07, Rev. 1, Section 3.12 to identify "safety analyses" (and thus safety analysis results). Furthermore, if the term "different result" were not limited to an examination of the results in the safety analyses, it is unclear which other results licensees would need to examine to satisfy *criterion 6. With the exception as stated in DG-1356, Section C.2.e, and without reasonable limits on which "different results" licensees should focus on, the NRC staff would be inviting the return of the uneven application of 10 CFR 50.59 that the 1999 amendment was intended to cure.*

To the extent that DG-1356, Section C.2.e argues that NEI 96-07, Appendix D, Section 4.3.6 reads the phrase "FSAR (as updated)" out of criterion 6 and, instead, replaces that phrase with "safety analysis," NEI disagrees. As previously explained, the focus on "safety analysis" within Section 4.3.6 is not based on the phrase "FSAR (as updated)," but rather is based on the phrase "different result." The question thus is where in the FSAR (as updated) are the "results" that were previously evaluated? Again, NEI submits that is reasonable to interpret "results" as "safety analysis results." In accordance with Definition 3.12, "Safety analyses are required to be presented in the UFSAR," and in alignment with the portion of 10 CFR 50.59(c)(2)(vi) that states, "any previously evaluated in the final safety analysis report (as updated)," NEI agrees that licensees must take a broad look at the UFSAR to identify any safety analyses that meet Definition 3.12. This examination is expressly not limited to specific sections of the UFSAR, instead licensees must take a wide view to determine which analyses or evaluations demonstrate that acceptance criteria for the facility's capability to withstand or respond to postulated events are met. Accordingly, safety analyses meeting Definition 3.12 may be found in any section of the UFSAR.

The NEI 96-07, Appendix D, Section 4.3.6 focus on the safety analyses meeting Definition 3.12, wherever they may be found in the UFSAR, is consistent with other 10 CFR 50.59 evaluation criteria and the guidance in NEI 96-07, Rev. 1. For example, 10 CFR 50.59(c)(2)(iii) considers accident consequences "previously evaluated in the final safety analysis report (as updated)." Notwithstanding an identical reference to the FSAR (as updated), it is well understood that this criterion is focused on safety analyses. Several 10 CFR 50.59 evaluation criteria utilize this logic with Definition 3.12 safety analyses as the focus and have done so since the 1999 rulemaking on 10 CFR 50.59. If the NRC staff proceeds with the exception as stated in DG-1356, Section C.2.e, it will reinstate the focus on the UFSAR wording rather than the various design functions and introduce inconsistent application among the 10 CFR 50.59 evaluation criteria.

Based on the NRC public meeting held on June 25, 2019, we agree that there are additional examples that could be included in NEI 96-07, Appendix D, Section 4.3.6 to illustrate cases that "create a possibility for a malfunction of an SSC important to safety with a different result." Attachment 1 provides proposed examples 4-23 and 4-24 based on the NRC's public meeting presentation examples of an emergency diesel generator voltage regulator control system and pressurizer power operated relief valves to control reactor coolant system pressure during low temperature operations. Incorporation of these examples in NEI 96-07, Appendix D, Section 4.3.6 as part of NRC's resolution of public comments should reassure NRC staff and licensees that the intent of the guidance appropriately captures the intent of 10 CFR 50.59(c)(2)(vi) consistent with NEI 96-07, Rev. 1.

<u>NRC Response</u>

This comment has been superseded by a revision to Appendix D submitted by NEI on May 13, 2020 (ADAMS Accession No. ML20129J857), to address the issues described in Section C.2.e of the draft regulatory guide. The NRC staff reviewed the May 13, submittal and found the revised Appendix D, Section 4.3.6 acceptable. As a result, a change was made to delete the exception in Section C.2.e of the draft regulatory guide.